

ATTACHMENT TO LICENSE AMENDMENT NO. 32

TO FACILITY COMBINED LICENSE NO. NPF-93

DOCKET NO. 52-027

Replace the following pages of the Facility Combined License No. NPF-93 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Facility Combined License No. NPF-93

REMOVE

7
8
11
12

INSERT

7
8
11
12

- (b) SCE&G shall report any violation of a requirement in Section 2.D.(3), Section 2.D.(4), Section 2.D.(5), and Section 2.D.(6) of this license within 24 hours. Initial notification shall be made to the NRC Operations Center in accordance with 10 CFR 50.72, with written follow up in accordance with 10 CFR 50.73.

(8) Incorporation

The Technical Specifications, Environmental Protection Plan, and ITAAC in Appendices A, B, and C, respectively of this license, as revised through Amendment No. 32, are hereby incorporated into this license.

(9) Technical Specifications

The technical specifications in Appendix A to this license become effective upon a Commission finding that the acceptance criteria in this license (ITAAC) are met in accordance with 10 CFR 52.103(g).

(10) Operational Program Implementation

SCE&G shall implement the programs or portions of programs identified below, on or before the date SCE&G achieves the following milestones.

- (a) Environmental Qualification Program implemented before initial fuel load;
- (b) Reactor Vessel Material Surveillance Program implemented before initial criticality;
- (c) Preservice Testing Program implemented before initial fuel load;
- (d) Containment Leakage Rate Testing Program implemented before initial fuel load;
- (e) Fire Protection Program
 - 1. The fire protection measures in accordance with Regulatory Guide (RG) 1.189 for designated storage building areas (including adjacent fire areas that could affect the storage area) implemented before initial receipt of byproduct or special nuclear materials that are not fuel (excluding exempt quantities as described in 10 CFR 30.18);
 - 2. The fire protection measures in accordance with RG 1.189 for areas containing new fuel (including adjacent areas where a fire could affect the new fuel) implemented before receipt of fuel onsite;

3. All fire protection program features implemented before initial fuel load;
- (f) Standard Radiological Effluent Controls implemented before initial fuel load;
 - (g) Offsite Dose Calculation Manual implemented before initial fuel load;
 - (h) Radiological Environmental Monitoring Program implemented before initial fuel load;
 - (i) Process Control Program implemented before initial fuel load;
 - (j) Radiation Protection Program (RPP) (including the ALARA principle) or applicable portions as identified in FSAR Section 12.5 thereof:
 1. RPP features applicable to receipt of by-product, source, or special nuclear materials (excluding exempt quantities as described in 10 CFR 30.18) implemented before initial receipt of such materials;
 2. RPP features (including the ALARA principle) applicable to new fuel implemented before receipt of initial fuel on site;
 3. All other RPP features (including the ALARA principle) except for those applicable to control radioactive waste shipment implemented before initial fuel load;
 4. RPP features (including the ALARA principle) applicable to radioactive waste shipment implemented before first shipment of radioactive waste;
 - (k) Reactor Operator Training Program implemented 18 months before the scheduled date of initial fuel load;
 - (l) Motor-Operated Valve Testing Program implemented before initial fuel load;
 - (m) Initial Test Program (ITP)
 1. Component Test Program implemented before the first component test;
 2. Preoperational Test Program implemented before the first preoperational test; and
 3. Startup Test Program implemented before initial fuel load;

9. The ITP procedures identified in FSAR Section 14.2.3
 - a. administrative manual (before the first component test)
 - b. preoperational testing (before scheduled performance)
 - c. startup testing (before initial fuel load)
- (f) Before initial fuel load, SCE&G shall:
 1. Update the seismic interaction analysis in AP1000 DCD, Rev. 19, Section 3.7.3.5 to reflect as-built information, which must be based on as-procured data, as well as the as-constructed condition;
 2. Reconcile the seismic analyses described in Section 3.7.2 of the AP1000 DCD, Rev. 19, to account for detailed design changes, including, but not limited to, those due to as-procured or as-built changes in component mass, center of gravity, and support configuration based on as-procured equipment information;
 3. Calculate the instrumentation uncertainties of the actual plant operating instrumentation to confirm that either the design limit departure from nucleate boiling ratio (DNBR) values remain valid or that the safety analysis minimum DNBR bounds the new design limit DNBR values plus DNBR penalties;
 4. Update the pressure-temperature (P-T) limits using the pressure temperature limits report (PTLR) methodologies approved in AP1000 DCD, Rev. 19, using the plant-specific material properties or confirm that the reactor vessel material properties meet the specifications of and use the Westinghouse generic PTLR curves;
 5. Verify that plant-specific belt line material properties are consistent with the properties given in AP1000 DCD Rev. 19, Section 5.3.3.1 and Tables 5.3-1 and 5.3-3. The verification must include a pressurized thermal shock (PTS) evaluation based on as-procured reactor vessel material data and the projected neutron fluence for the plant design objective. Submit this PTS evaluation report to the Director of NRO, or the Director's designee, in writing, at least 18 months before initial fuel load;
 6. Review differences between the as-built plant and the design used as the basis for the AP1000 seismic margin analysis. SCE&G shall perform a verification walkdown to identify differences between the as-built plant and the design. SCE&G shall evaluate any differences and must modify the seismic margin analysis as necessary to account for the plant-specific design and any design

changes or departures from the certified design. SCE&G shall compare the as-built structures, systems, and components (SSC) high confidence, low probability of failures (HCLPFs) with those assumed in the AP1000 seismic margin evaluation, before initial fuel load. SCE&G shall evaluate deviations from the HCLPF values or assumptions in the seismic margin evaluation due to the as-built configuration and final analysis to determine if vulnerabilities have been introduced;

7. Review differences between the as-built plant and the design used as the basis for the AP1000 probabilistic risk assessment (PRA) and the AP1000 DCD, Rev. 19, Table 19.59-18. SCE&G shall evaluate the plant-specific PRA-based insight differences and shall modify the plant-specific PRA model as necessary to account for the plant-specific design and any design changes or departure from the PRA certified in Rev. 19 of the AP1000 DCD;
8. Review differences between the as-built plant and the design used as the basis for the AP1000 internal fire and internal flood analysis. SCE&G shall evaluate the plant-specific internal fire and internal flood analyses and shall modify the analyses as necessary to account for the plant-specific design and any design changes or departures from the design certified in Rev. 19 of the AP1000 DCD; and
9. Perform a thermal lag assessment of the as-built equipment listed in Tables 6b and 6c in Attachment A of APP-GW-GLR-069, "Equipment Survivability Assessment," to provide additional assurance that this equipment can perform its severe accident functions during environmental conditions resulting from hydrogen burns associated with severe accidents. SCE&G shall perform this assessment for equipment used for severe accident mitigation that has not been tested at severe accident conditions. SCE&G shall assess the ability of the as-built equipment to perform during accident hydrogen burns using the environment enveloping method or the test based thermal analysis method described in Electric Power Research Institute (EPRI) NP-4354, "Large Scale Hydrogen Burn Equipment Experiments."
10. Implement a surveillance program for explosively actuated valves (squib valves) that includes the following provisions in addition to the requirements specified in the edition of the ASME *Code for Operation and Maintenance of Nuclear Power Plants* (OM Code) as incorporated by reference in 10 CFR 50.55a.